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FAST FACILITY SPENT-FUEL AND WASTE ASSAY INSTRUMENT[a]

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ABSTRACT

A delayed-neutron assay instrument was installed in the Fluorinel Dissolution and Fuel Storage Facility at Idaho National Engineering Laboratory. The dual-assay instrument is designed to measure both spent fuel and waste solids that are produced from fuel processing. A set of waste standards, fabricated by Los Alamos using uranium supplied by EXXON Nuclear Idaho Company, was used to calibrate the small-sample assay region of the instrument. Performance testing was completed before installation of the instrument to determine the effects of uranium enrichment, hydrogenous materials, and neutron poisons on assays. The unit was designed to measure high-enriched uranium samples in the presence of large neutron backgrounds. Measurements indicate that the system can assay low-enriched uranium samples with moderate backgrounds if calibrated with proper standards.

INTRODUCTION

A delayed-neutron interrogation assay system (Shuffler) was designed, constructed, and installed in the new Department of Energy Fluorinel Dissolution and Fuel Storage (FAST) Facility at Idaho National Engineering Laboratory (INEL).[1-3] The Shuffler, developed as an integral part of the FAST Facility design, evolved through close interactions among Los Alamos National Laboratory, EXXON Nuclear Idaho Company (ENICO), and the Ralph M. Parsons Company, the architectural engineers. Installation of the instrument at the facility was coordinated through Catalytic, Inc., the construction manager. The facility will store and reprocess uranium fuel from US Government research reactors and the US Navy's nuclear ship propulsion program.[4] The Shuffler is a dual-assay device designed to measure nondestructively the ^{235}U content in enriched-uranium spent fuels and in waste solids produced by fuel dissolution. Two independent estimates of the uranium content in fuels are required before dissolution to satisfy plant operating procedures and to ensure criticality safety during processing. These estimates are provided by a Shuffler assay measurement and an operator-declared burnup value. Following fuel dissolution and complexing, the solution is filtered to remove undissolved solids and then sent to another facility for uranium recovery. The insoluble particulate removed by the filter constitutes solids waste, which is packed into stainless steel canisters for eventual disposal. Material accountability requires a determination of the quantity of uranium contained in waste solids, a difficult problem when chemical techniques are used. The Shuffler is designed to measure the ^{235}U content in waste canisters.

Assay specifications for the instrument consist of a waste-canister measurement precision of ±30 g (2σ) over the range 0 to 400 g of ^{235}U. Specifications for fuel measurements require a ±5% (2σ) precision for fuel loadings between 2 and 10 kg of ^{235}U. Assay measurements are complicated by large neutron backgrounds produced from both (α,n) reactions and spontaneous fission of the curium isotopes. Operational specifications list a neutron background from the fuel and waste of up to 1.2×10^{7} s^{-1} and require the instrument to meet the measurement specifications for backgrounds up to 0.5×10^{6} s^{-1}.

This paper contains a brief description of the Shuffler assay instrument and includes a discussion of the software features used to perform assays and to provide graphical displays of calibration data and the uranium profiles in samples. Waste-canister standards developed to calibrate the instrument are discussed, and data on scanning assays of these canisters are presented. Measurement results are presented for a variety of materials assayed during performance testing of the instrument.

System Description

Because waste solids and spent-fuel components were not available for experiments, the

[a]Work supported by the US Department of Energy, Office of Safeguards and Security.

instrument was designed using analytical calculations.[2] The Los Alamos Monte Carlo code (MCNP)[5] was used to specify a three-dimensional geometry and simulate neutron transport in the instrument to determine fission and delayed-neutron count rates[6] for various samples. The types of materials and their geometries and locations in the instrument, in addition to assessment of the Shuffler's measurement performance, were based on MCNP calculations.[2]

The Shuffler (Fig. 1) is housed in a separate shielded cubicle at the base of the dissolver cell and contains two tubes that penetrate the cell floor and extend down through the instrument. These tubes provide a containment barrier between the contaminated process cell and the instrument and permit scanning measurements to be performed on samples passed through the Shuffler. The 15-cm-diam small-sample tube is for waste-canister assays and provides a geometry suitable for precision measurements on samples containing small quantities of uranium. The 50-cm-diam large-sample tube enables measurement of fuel packages that have various shapes and lengths and contain large quantities of uranium.

Quality-assurance and quality-control programs were developed and records were maintained during design, development, and testing of the Shuffler. The instrument is required to withstand earthquakes of a specified magnitude and frequency and remain operational. The Shuffler also satisfies safety, radiation shielding, and facility maintenance specifications. All components in the instrument are labeled, permitting the instrument to be disassembled and reassembled. Documentation supplied with the system consists of a software operations manual, an electronics systems manual, physics and measurement test documentation, and complete mechanical fabrication and electrical schematic drawings.

The assay system comprises both electronic and mechanical components that are located throughout the facility (Fig. 2). The assay instrument, contained in a cubicle, is connected by cables to nuclear electronics instrumentation mounted in an equipment rack outside the cubicle. A computer located in the operations room controls the instrument and receives measurement data from the nuclear electronics. The instrument is operated from a terminal by selecting an option from the assay software menu. This terminal is located adjacent to a hot-cell window, directly above the instrument cubicle, in the crane corridor. This location provides convenient access for operators to view the samples being positioned for measurement. Sample information and assay data are transferred between the Shuffler and main facility computer using a communications link. The vertical position of a sample in an assay tube during measurements is obtained directly by the Shuffler computer through a link with the hot-cell crane.

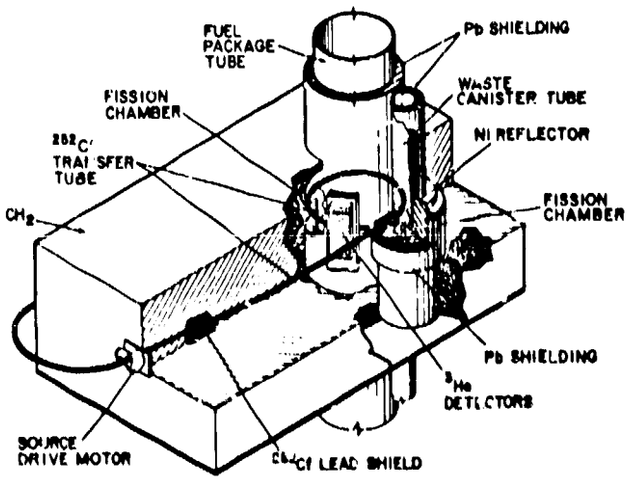


Fig. 1. Three-dimensional schematic of the FAST delayed-neutron interrogator.

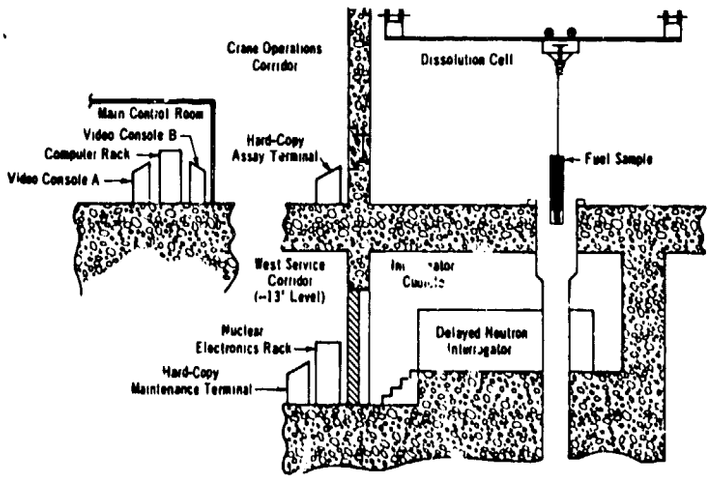


Fig. 2. Delayed-neutron interrogator instrument component locations in the FAST facility.

Shielding is required to reduce the radiation emitted from the waste and spent-fuel samples and from the ^{252}Cf source contained in the instrument. Radiation reaching the personnel corridor is reduced by the shield cubicle housing the instrument. During a measurement, the intense gamma radiation emitted from fuel and waste samples is also decreased by a 10-cm-thick lead column surrounding the sample tubes. This column consists of 350 interlocking lead pieces clamped into position by an external shroud. The Shuffler is shielded to reduce the number of gamma rays and neutrons emitted by the ^{252}Cf source that escape from the shield. The source is kept in the shield position by an interlock key switch that is used when the cubicle is open to personnel access. Radiation reaching the surface of the instrument is <5 mR/h for the largest allowable ^{252}Cf source (3.75 mg), except for one position where the radiation level is 25 mR/h. This radiation is reduced to a negligible quantity on the outside of the cubicle housing the Shuffler.

A source transfer cask was designed and constructed to permit ^{252}Cf sources to be transferred to and from the Shuffler. The cask meets the Department of Transportation (DOT-7A) license requirements and is certified to transport up to 2 Ci (3.7 mg) of ^{252}Cf. Dose measurements with a 0.82-mg ^{252}Cf source indicate that the largest allowable source would produce ~30 mR/h at the cask surface and 2 mR/h at 1 m from the cask. Special features of this cask are storage positions for two sources and a gear mechanism to transfer the sources. Each storage position is connected to a gear wheel, and a californium source attached to a Teleflex cable can be inserted or removed from the cask by connecting a hand crank to the appropriate gear. The shield cask allows sources to be quickly and easily transferred and provides radiation protection during removal of an old source and insertion of a new source into the Shuffler, as shown in Fig. 3.

The instrument exceeds the seismic design requirements for the facility. The Shuffler is rigidly bolted to a mounting pedestal in the cubicle. All materials are tightly positioned within a sturdy framework to prevent movement or shifting. The lead column surrounding the two tubes that penetrate the instrument cubicle is clamped in place by a stainless steel shroud, which is bolted to the pedestal base and the cubicle ceiling. During an earthquake, the shroud will limit movement of the lead pieces and prevent rupture of the tubes.

The assay procedure consists of lowering a sample into the appropriate through-tube for a background measurement, followed by an active neutron interrogation as the sample is lifted from the tube. Interrogation consists of transferring the ^{252}Cf source from its shielded storage position to an irradiation location near the sample. After irradiation the source is returned to storage and then delayed neutrons produced as a result of uranium fissions induced by the source are counted. This measurement

Fig. 3. A source is transferred from the Shuffler to the shield cask.

cycle is repeated until the sample is lifted from the measurement region of the Shuffler. Using the assay measurement and a calibration curve obtained from known standards, a value for the ^{235}U content in the sample is computed.

Performance Test Results

Performance testing was completed before installation of the Shuffler in the FAST facility. A ^{252}Cf source was obtained from Savannah River Laboratories and calibrated by the National Bureau of Standards (NBS). The source neutron emission rate was specified by NBS to be 1.76×10^9 n/s on January 1, 1982. Experiments were conducted using this source to determine the effects of hydrogen, poisons, and uranium enrichment on assays. Data were collected from a variety of samples consisting of low-enriched uranium in light-water-reactor (LWR) and boiling-water-reactor (BWR) fuel rods, high-enriched uranium contained in graphite powder, and materials-test-reactor (MTR) fuel plates. A set of waste standards was fabricated and used to calibrate the Shuffler small-sample assay region.

The detection efficiency and vertical response profiles were measured using a small calibrated ^{252}Cf source. The ^3He detection efficiency was measured with the source positioned at the detector midplane and centered in each assay tube. The efficiency of the small- and large-sample assay tubes was 7.4 and 1.2%. The vertical profile response was measured by collecting counts with the source set to various positions along the center of each assay tube. The vertical profiles are plotted in Figs. 4 and 5 for the small- and large-sample assay regions.

ENICO supplied the uranium and Los Alamos fabricated waste standards to calibrate the Shuffler. Material composing the standards was prepared using information supplied by ENICO on the quantities expected in waste solids from fuel processing. Four canisters with identical shapes and constructed of stainless steel 304 were supplied by ENICO. The canisters have a 0.32-cm wall thickness, 11.43-cm o.d., 61-cm height, and an internal volume of 4.65 ℓ. Material was accurately weighed and mixed using a V-blender. During filling operations, samples were taken at various positions in the canister for analysis of boron, cadmium, uranium, and water content and also to obtain an estimate of the uniformity of mixing. Table I lists the gram quantities of materials in each waste canister. Standards 1, 3, and 4 were fabricated with sufficient matrix material to fill them. Canister 2 was fabricated with only enough matrix material to fill it approximately half-way.

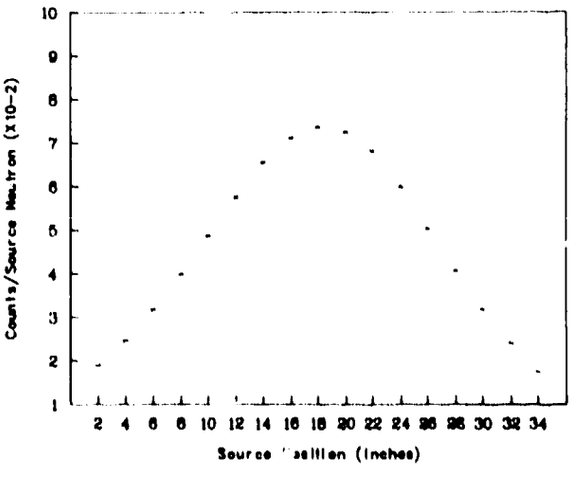


Fig. 4. Vertical response profile measured with a ^{252}Cf source positioned along the center line of the small-sample assay tube.


CAUTION
RADIOACTIVE MATERIAL
KEEP OUT
RADIATION AREA

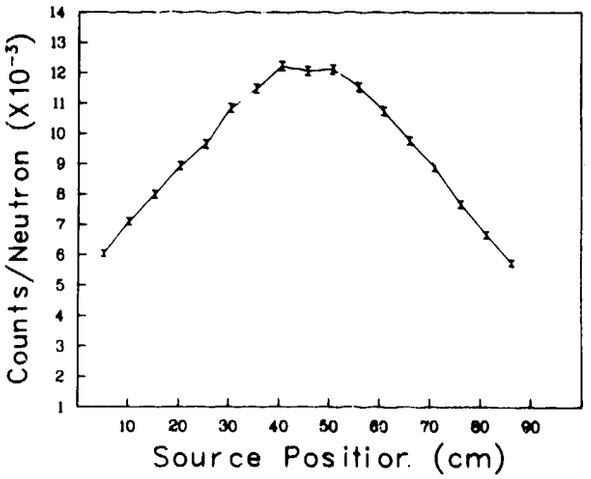


Fig. 5. Vertical response profile measured with a ^{252}Cf source positioned along the center line of the large-sample assay tube.

TABLE I

WASTE-CANISTER MATERIALS[a]

Canister Number	ZrO_2 (g)	B_4C (g)	CdO (g)	UO_3 (g)	Total (g)	^{235}U (g)
1	10 068	311	105	84.9	10 568.9	49.5
2	5 017	155	53	170	5 395	99.0
3	10 113	313	105	341	10 872	198.6
4	10 114	312	106	682	11 214	397.3

[a]The weights of materials in each canister are based on the assumption that the blend is homogeneous and the materials are pure.

A partially filled canister was needed to study the effects of fill height on waste assays.

Small-Sample Assay Tube Measurements

Assay scanning measurements were completed on the four waste standards. The 50.8-cm-long samples were lowered and raised through the small tube at a constant rate of 12.7 cm/min. Measurements were continued while the sample was moving through the 76-cm-long scanning region. This region was selected based on the vertical response profile of the delayed-neutron detectors. Background data were collected from each sample during the down scan, and a cyclical ^{252}Cf irradiation delayed-neutron measurement sequence was completed during the up scan. The cyclical sequence consisted of an 8-s sample irradiation followed by a 4-s delayed-neutron count. Approximately 0.5 s was required to move the source between the sample irradiation and storage positions. During the irradiation sequence, neutron counts were collected using low-efficiency ^{235}U flux-monitor detectors. These detectors, near the sample, monitor the source irradiation strength, fission rate, and hydrogen content of the sample. When the source is returned to storage, delayed-neutron counts are collected using ^{3}He detectors. During the scanning process, between 36 and 38 irradiation counting cycles were completed on each canister. Variability in the number of cycles resulted from slight changes in the scan rate for each canister. Table II lists the count rates collected from the waste canisters corresponding to the ^{252}Cf irradiation source strength equal to 1.41×10^9 n/s on November 3, 1982, the measurement date. Figure 6 is a plot of the measured waste standard data. The points fit a straight line indicating that self-shielding and multiplication effects are negligible. A fit to the equation $R = B\,M$, where R is the measured response and M is the mass of the sample, produced a value for the slope coefficient (B) of 2.33.

Repeatability measurements were conducted with waste standard 4, containing 397.3 g of ^{235}U, in a fixed position at the center of the small-sample assay tube. Fifty irradiation-counting cycles were completed for each measurement, and a wait time of 10 min was imposed between measurements to allow the delayed-neutron precursors to decay. Each cycle required 13.0 s to complete and consisted of a 0.5-s source transfer time, an 8.0-s irradiation time, a 0.5-s source transfer time, and a 4.0-s delayed-neutron count time. Movement of the source between the storage and sample irradiation positions requires a transfer time of ~0.5 s. The fixed-position repeatability assay data for this

TABLE II

WASTE-CANISTER SCANNING ASSAY DATA
(11/3/82)

Canister Number	^{235}U (g)	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM[a] Ratio
1	49.5	117.4 ± 1.2	1648.1 ± 2.1	0.071
2	99.0	230.1 ± 1.5	1668.3 ± 2.3	0.138
3	198.6	463.5 ± 1.9	1637.4 ± 2.2	0.283
4	397.3	914.4 ± 2.6	1639.8 ± 2.2	0.558

[a]DN/FM is the ratio of the delayed-neutron-to-flux-monitor count rates.

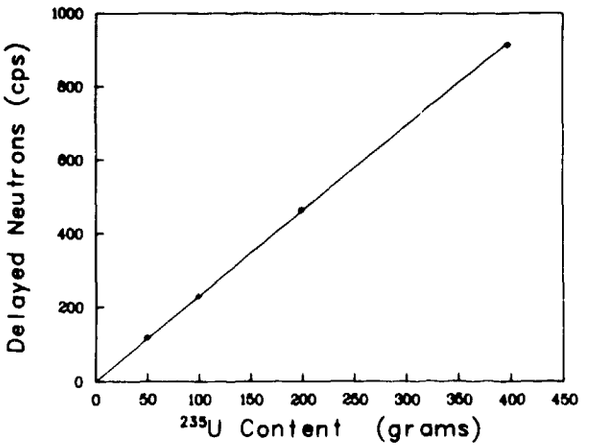


Fig. 6. Delayed-neutron detector response vs the ^{235}U mass for the four waste standards measured in the small-sample assay tube.

sample are listed in Table III and plotted in Fig. 7. Delayed-neutron count rates are significantly higher for a stationary sample at the center of the assay region compared with scanning measurements. A scanning assay will have a lower delayed-neutron rate because of end effects as the sample is moved through the measurement region of the instrument. A background was collected for each measurement and subtracted from the delayed-neutron and flux-monitor counts. Background rates averaged 74.3 ± 0.4 counts/s and 0.05 ± 0.01 counts/s in the delayed-neutron and flux-monitor detectors, respectively, for the 18 repeatability measurements listed in Table III. The arithmetic average and standard deviation of the 18 measurements are listed at the bottom of Table III. The repeat measurement precisions for the delayed-neutron and flux-monitor count rates in this test were 0.18 and 0.12%, respectively. The average delayed-neutron-to-flux-monitor ratio was 1.295 ± 0.002 with a standard deviation of 0.17%. The standard deviation of the delayed-neutron and flux-monitor data is in close agreement with the Poisson statistics listed for each measurement. Variabilities in the instrument could increase the standard deviation above the Poisson statistics. For example, if the source transfer times were inconsistent or the source position for sample irradiation varied significantly between cycles, then the repeat measurement precision could be much larger than indicated by the Poisson statistics from a given measurement. The agreement between repeat assays is within the uncertainty

TABLE III

REPEATABILITY ASSAYS ON WASTE STANDARD 4

Measurement Number	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM[a] Ratio
1	1977.0 ± 3.2	1528.9 ± 1.8	1.293
2	1980.4 ± 3.2	1531.8 ± 1.8	1.293
3	1978.4 ± 3.2	1529.4 ± 1.8	1.294
4	1976.1 ± 3.2	1528.8 ± 1.8	1.293
5	1974.2 ± 3.2	1527.6 ± 1.8	1.292
6	1983.6 ± 3.2	1527.9 ± 1.8	1.298
7	1975.2 ± 3.2	1528.7 ± 1.8	1.292
8	1982.0 ± 3.2	1528.7 ± 1.8	1.297
9	1980.4 ± 3.2	1529.6 ± 1.8	1.295
10	1975.1 ± 3.2	1528.8 ± 1.8	1.292
11	1979.4 ± 3.2	1527.7 ± 1.8	1.296
12	1981.2 ± 3.2	1532.4 ± 1.8	1.293
13	1981.9 ± 3.2	1530.8 ± 1.8	1.295
14	1979.4 ± 3.2	1526.9 ± 1.8	1.296
15	1977.2 ± 3.2	1524.3 ± 1.8	1.297
16	1983.8 ± 3.2	1530.1 ± 1.8	1.297
17	1979.2 ± 3.2	1526.5 ± 1.8	1.297
18	1973.0 ± 3.2	1528.8 ± 1.8	1.291
Mean	1978.5 ± 3.6	1528.8 ± 1.9	1.295 ± 0.002

[a]DN/FM is the ratio of the delayed-neutron-to-flux-monitor count rates

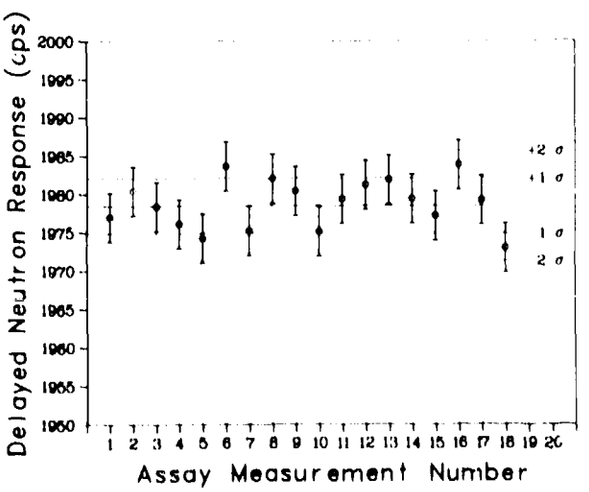


Fig. 7. Repeatability measurements on the 397.3-g ^{235}U waste standard (number 4) at a fixed position centered in the small-sample assay tube.

determined using Poisson statistics, indicating that instrument variability is not a significant source of error for measurements on fixed samples.

Measurements were completed on a set of four high-enriched uranium graphite standards packaged in polyethylene bottles and located in a fixed position in the center of the small-sample assay tube. The uranium was enriched to 92.83% and uniformly mixed in graphite powder. Table IV lists the data collected from a 40-cycle measurement sequence on each sample. The effects of neutron moderation by the graphite and the resulting uranium sample self-shielding produce a decreasing response as the uranium loading is increased. The shape of the curve (Fig. 8), because of the shielding effect, is not linear and bends down with increasing mass.

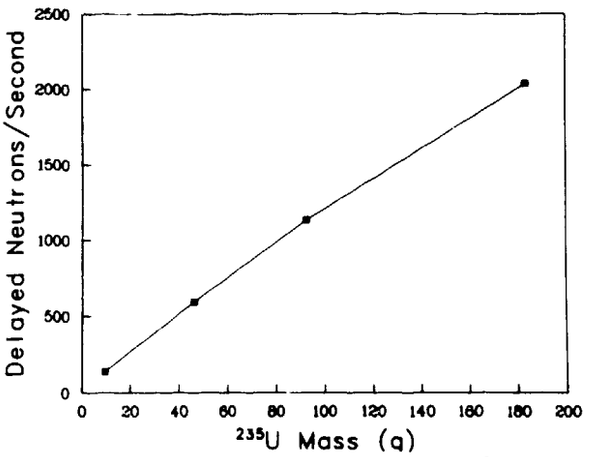


Fig. 8. Delayed-neutron count rate vs ^{235}U mass for assay of high-enriched uranium-graphite samples.

Moderation- and Poison-Effect Measurements

The effects of hydrogenous and poison materials in samples being measured were studied. Three sets of samples were prepared using a V-blender. The first set consisted of zirconium-oxide powder blended with four different quantities of sugar. The sugar provided hydrogen and was mixed to form samples with approximately 0.5, 1.0, 1.5, and 2.0 wt% hydrogen in the mix. The second and third sets of standards were identical to the first set except that 2.0 and 3.5 wt% boron carbide was added to the mix. Each measurement sample comprised four stacked, thin-wall, aluminum cans having a total height of 55 cm and a diameter of 10.8 cm, which is similar to the size of the waste canisters. Four PWR Zircaloy fuel tubes cut to 7.6-cm lengths were filled with the UO_3 remaining from the waste standard fabrication. Each of the tubes, containing 11.26, 10.94, 10.87, and 10.91 g of ^{235}U, was inserted down the center line of an aluminum can before measurement. The cans were then stacked and scanned through the sample-assay tube. Tables V, VI, and VII list the data collected from the three sample sets. The delayed-neutron-to-flux-monitor ratios for the three sets are plotted in Fig. 9. Increasing the hydrogenous material increases the

TABLE IV

URANIUM-GRAPHITE SAMPLE ASSAY DATA

Sample ID	^{235}U (g)	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM Ratio
434	9.283	138.7 ± 1.2	2097.8 ± 2.4	0.066
416	46.42	595.0 ± 2.1	2068.8 ± 2.4	0.288
417	92.83	1135.4 ± 2.8	2052.6 ± 2.4	0.553
418	185.66	2038.5 ± 3.7	1981.8 ± 2.3	1.029

TABLE V

SMALL-SAMPLE ASSAY TUBE MODERATION DATA
(No Poison Materials)

Hydrogen (wt%)	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM[a] Ratio
0.0	108.8 ± 1.1	1717.4 ± 2.1	0.0634
0.467	194.1 ± 1.4	2183.3 ± 2.5	0.0889
0.885	276.2 ± 1.5	2472.8 ± 2.5	0.1117
1.24	393.2 ± 1.7	2771.3 ± 2.7	0.1419
1.56	474.0 ± 1.8	2955.5 ± 2.8	0.1604

[a]DN/FM is the ratio of the delayed-neutron counts to the flux-monitor counts.

TABLE VI

SMALL SAMPLE-ASSAY TUBE MODERATION PLUS POISON DATA
(2.0 wt% Boron Carbide)

Hydrogen (wt%)	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM Ratio
0.0	108.5 ± 1.1	1708.4 ± 2.1	0.0635
0.47	149.4 ± 1.2	1972.0 ± 2.4	0.0758
1.56	235.9 ± 1.4	2536.4 ± 1.8	0.0930

TABLE VII

SMALL-SAMPLE ASSAY TUBE MODERATION PLUS POISON DATA
(3.5 wt% Boron Carbide)

Hydrogen (wt%)	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM[a] Ratio
0.0	108.5 ± 1.1	1708.4 ± 2.1	0.0635
0.452	126.8 ± 1.1	1863.6 ± 2.2	0.0680
0.850	150.8 ± 1.2	2057.9 ± 2.4	0.0733
1.20	172.8 ± 1.3	2253.5 ± 2.5	0.0767
1.56	199.9 ± 1.3	2372.9 ± 2.5	0.0842

[a]DN/FM is the ratio of the delayed-neutron counts to the flux-monitor counts.

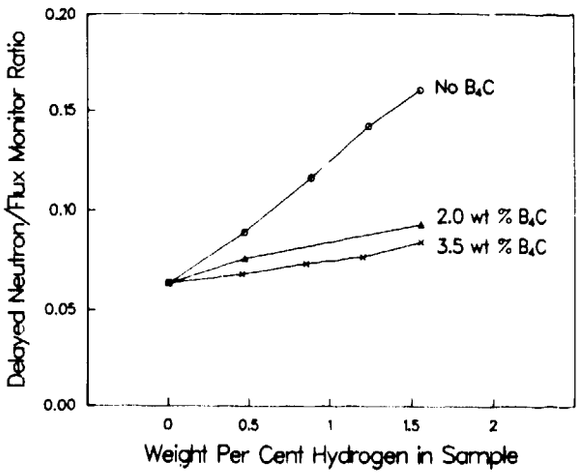


Fig. 9. Delayed-neutron-to-flux monitor ratio vs weight per cent hydrogen in samples containing a fixed quantity of uranium with boron poison loadings of 0, 2.0, and 3.5 wt%.

delayed-neutron response from a fixed quantity of ^{235}U in a sample. The flux-monitor response also increases with the quantity of hydrogen moderator. Using the ratio of delayed-neutron counts to flux-monitor counts helps decrease the measurement sensitivity to the hydrogen content in a sample, but as shown in Fig. 9, measurements are still affected by the hydrogen loading. Adding neutron poison to the sample decreases the number of thermal neutrons in the sample and aids in further decreasing the effects of hydrogenous materials on a sample assay. Waste canisters to be measured by the FAST delayed-neutron interrogator will be dried to remove the unbound hydrogen and are expected to contain hydrogen loadings between 0 and 1.0 wt% and poison loadings between 3.0 and 4.0 wt%.

Uranium-Enrichment-Effect Measurements

A series of measurements was completed in the small- and large-sample assay tubes with 1-kg uranium standards containing ^{235}U enrichments between 1.94 and 91.32%. The measured data are listed in Table VIII. The delayed-neutron response increased, and the flux-monitor response remained essentially the same for a given uranium loading as the enrichment increased. However, the delayed-neutron response per gram decreased with increasing enrichment, as shown in Fig. 10, because of an increasing delayed-neutron response from ^{238}U fissions as the enrichment was decreased. The response remained relatively flat for enrichments >30%. Measurements of low-enriched materials require a calibration curve using standards that match the enrichments of the unknown materials. The response from ^{238}U fissions could be decreased by moderating the ^{252}Cf neutron flux. However, the sample penetrability would be decreased and measurements on poison-loaded samples would be more difficult.

Large-Sample Assay Tube Measurements

The large-sample assay tube is designed to measure samples with large uranium loadings. The ^{252}Cf irradiation source circles this tube to provide a more uniform sample interrogation than that from irradiating with the source at a

TABLE VIII

URANIUM ENRICHMENT ASSAY DATA
(1000-g Standards at Various Enrichments)

Sample ID	Enrichment (wt%)	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM Ratio
UISO-2	1.94	1016.9 ± 3.7	1620.6 ± 3.0	0.627
UISO-3	3.02	1086.0 ± 3.9	1615.1 ± 3.0	0.673
UISO-10	10.06	1725.5 ± 4.8	1588.7 ± 3.0	1.086
UISO-12	11.77	1850.1 ± 4.9	1589.1 ± 3.0	1.164
UISO-13	13.31	1913.8 ± 5.0	1584.2 ± 3.0	1.208
UISO-17	17.34	2000.1 ± 5.1	1592.9 ± 3.0	1.256
UISO-27	26.84	2636.8 ± 5.9	1577.7 ± 2.9	1.671
UISO-38	37.70	3237.8 ± 6.5	1572.0 ± 2.9	2.060
UISO-52	52.21	3899.7 ± 7.1	1573.8 ± 2.9	2.478
UISO-66	66.14	4331.6 ± 7.4	1573.8 ± 2.9	2.752
UISO-91	91.32	6178.8 ± 8.9	1553.9 ± 2.9	3.976

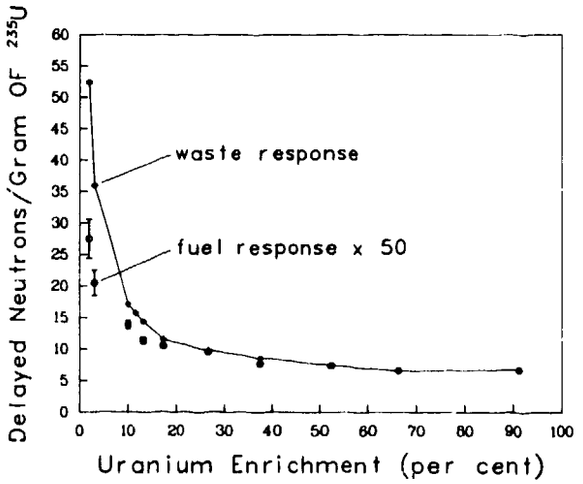


Fig. 10. Small-sample assay region measurements of delayed-neutron response per gram of ^{235}U vs uranium enrichment.

fixed position. Initial experiments were conducted by scanning the waste canisters through the large-sample assay tube. The assay procedure is similar to that of the small-sample assay tube. A background is collected during the sample down scan, and the irradiation and delayed-neutron counting cyclical sequence are performed during the sample up scan. The irradiation sequence used to measure the waste standards in the large assay tube consisted of a 0.5-s source transfer, an 8.0-s irradiation while the source was moved slowly around the assay tube, a 0.5-s source transfer, and a 4.0-s delayed-neutron count. The measurement data, listed in Table IX, show a linear response for the uranium loadings in the waste-material matrix. These experiments indicate that the large-sample assay region is capable of measurements on samples containing low uranium loadings when the background levels are also low. Comparison of the waste-standard measurements between the small- and large-sample assay regions shows, as expected, that better measurement precision is obtained using the small-sample assay tube.

Scanning assay measurements on a 6 x 6 array of BWR fuel rods were collected using the large-sample assay tube. The 180-cm-long rods were scanned at a rate of 12.7 cm/min. Measurements were collected with the BWR array containing from 8 to 36 fuel rods that have a ^{235}U enrichment of 2.36%. The BWR measurement data

TABLE IX

LARGE-SAMPLE-REGION WASTE-CANISTER SCANNING DATA
(9/18/82)

Sample ID	^{235}U (g)	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM Ratio
1	49.5	3.77 ± 0.95	1186.3 ± 2.1	0.0032
2	99.0	10.43 ± 0.98	1190.5 ± 2.1	0.0088
3	198.6	19.86 ± 1.04	1187.8 ± 2.1	0.0167
4	397.3	38.90 ± 1.10	1186.8 ± 2.1	0.0328

(Table X), plotted in Fig. 11, show a linear response with fuel-pin loading.

Fixed-position assay data were collected on a 15 x 15 array of PWR fuel rods that had enrichments of 3.19%. The fuel array was positioned down the center of the large-sample assay tube and centered at the detector's midplane. Data were collected for 20 cycles using an 8.0-s irradiation time and a 4.0-s delayed-neutron count time. The PWR measurement data for a series of rod loadings are listed in Table XI. Figure 12 is a plot of the PWR data and shows a linear response with fuel-pin loading.

Software Features

Control of the assay system is automated by software commands entered at the terminal located in the crane corridor. For example, typing the letter A at the terminal initiates an assay sequence; the Shuffler then obtains all information required for the measurement, such as sample identification and assay location, directly from the facility computer. After information is received, the operator is instructed to position the sample and lower it into the assay tube for measurement. The software provides simple, clear messages listing problems and directing actions required of operators. When problems appear, the operator can override the system and continue measurements;

TABLE X

BWR FUEL ROD SCANNING ASSAY DATA

No. of Rods	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM Ratio
8	84.99 ± 1.53	1149.3 ± 2.5	0.074
12	126.06 ± 1.69	1154.1 ± 2.5	0.109
18	193.83 ± 1.92	1151.4 ± 2.5	0.168
24	255.24 ± 2.12	1159.6 ± 2.5	0.220
36	374.10 ± 2.45	1162.6 ± 2.5	0.322

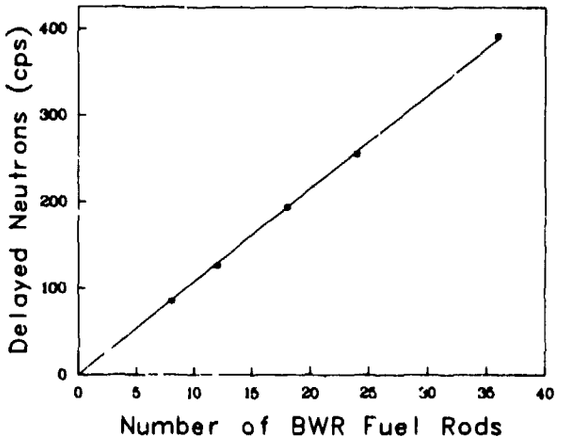


Fig. 11. Large-assay-region scanning data from BWR fuel rods.

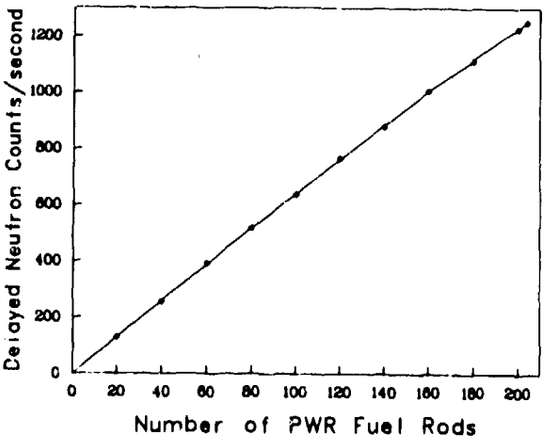


Fig. 12. Large-assay-region fixed-sample data from PWR fuel rods.

TABLE XI

LARGE-SAMPLE-TUBE PWR ASSAY DATA

No. of Rods	Delayed Neutron (DN) (counts/s)	Flux Monitor (FM) (counts/s)	DN/FM Ratio
20	128.8 ± 1.7	1146.1 ± 2.5	0.112
40	255.5 ± 2.1	1148.3 ± 2.5	0.223
60	391.8 ± 2.5	1149.2 ± 2.5	0.341
80	516.3 ± 2.8	1148.8 ± 2.5	0.449
100	636.8 ± 3.1	1149.4 ± 2.5	0.554
120	764.0 ± 3.3	1156.0 ± 2.5	0.661
140	877.6 ± 3.5	1157.0 ± 2.5	0.759
160	1005.2 ± 3.8	1163.0 ± 2.5	0.864
180	1111.8 ± 3.9	1162.3 ± 2.5	0.957
200	1228.6 ± 4.1	1162.7 ± 2.5	1.057
204	1251.1 ± 4.2	1164.2 ± 2.5	1.075

however, all data collected are flagged when a problem is detected. The software is designed to be user friendly and to satisfy facility operational requirements for assays performed by minimally trained personnel on a production schedule.

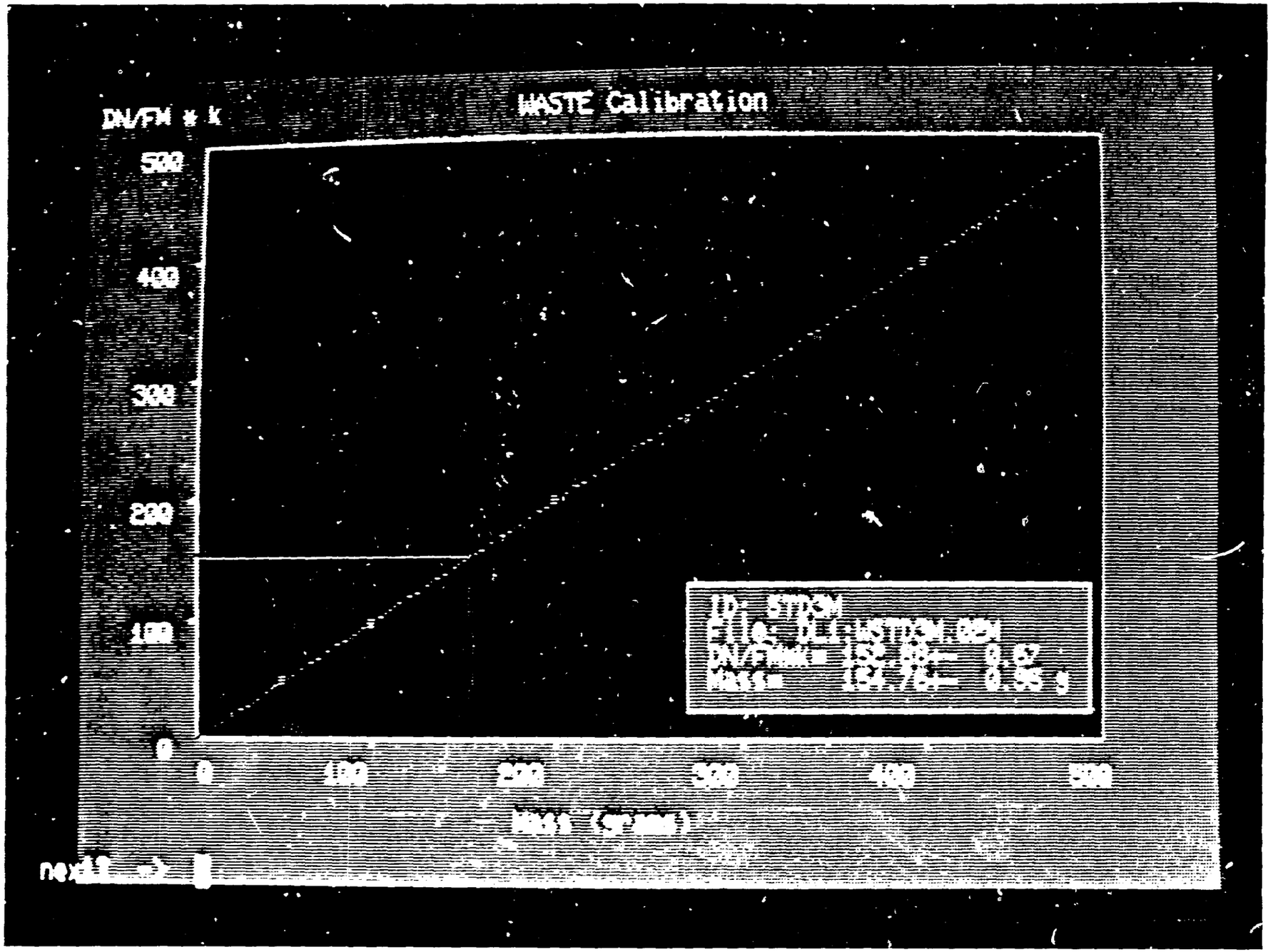
We developed three graphics programs to show additional information to facility personnel. One color graphics program simulates instrument operation and shows movement of the source, sequence of operation of the delayed-neutron and flux-monitor detectors, and placement of samples for assay in either the small- or large-sample assay regions. A graphics calibration program plots the assay data points and measurement uncertainties from standards vs the uranium mass of a sample. Figure 13 is a plot of the small-sample assay region calibrated with the waste standards, which shows the calibration curve and the interpolated assay value from a measurement of an unknown sample. The profile of the uranium distribution along the length of a sample is obtained as a result of scanning. A color graphics program showing the profile distribution from a scanning measurement is illustrated in Fig. 14. The data in this plot corresponded to scanning a waste canister filled with zirconium oxide. The canister contained a 200-g ^{235}U disk placed at the top with a 40-g uranium foil placed on the bottom. The graphics programs provide other information in addition to the uranium assay value.

Conclusion

The Shuffler assay system is installed at the FAST facility. Connections to the facility crane and main facility computer will be completed by December 1983. Testing of the system with cold fuel components is scheduled for late 1984. A training course will be taught at the facility to orient operators and supervisors on use of the Shuffler for scanning assays of both waste and fuel samples. Based on initial testing with fabricated waste standards, the instrument can obtain measurements with precisions of well under 1.0%. The addition of neutron poisons to waste materials counteracts the effects of hydrogen loadings on measurements. Using the ratio of delayed-neutron counts to flux-monitor

Fig. 13. Waste-standard calibration data vs uranium mass for measurements in the small-sample assay tube.

Fig. 14. Profile plot of the uranium distribution in a sample determined from a scanning measurement in the small-sample assay tube. The 50 cm-long sample was filled with zirconium oxide and contained a 200-g disk of ^{235}U on the top and 40-g of ^{235}U on the bottom.


WASTE Calibration
DN/FM * K
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400
300
200
100
0
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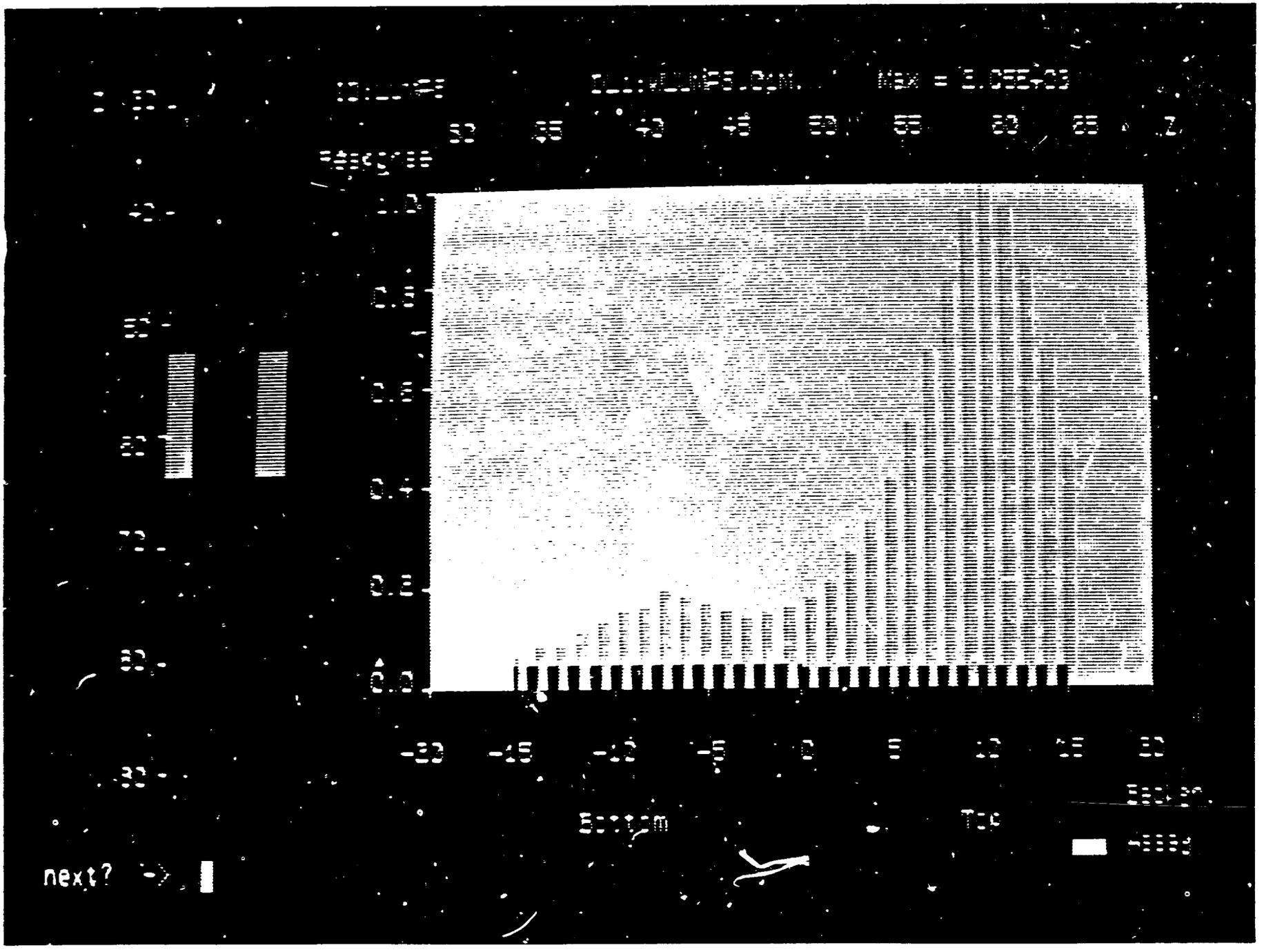
100
200
300
400
500


Max = 1.0E+03
1.0
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counts decreases the effect of hydrogen moderator on uranium measurements of waste samples. The accuracy of assays at the facility will depend on the background neutron level from radioactive samples and will require development of standards for instrument assay calibration that are representative of the materials in the process.

REFERENCES

1. H. O. Menlove, G. Eccleston, D. A. Close, and L. G. Speir, "Neutron Interrogator Assay System for the Idaho Chemical Processing Plant Waste Canisters and Spent Fuel - Preliminary Description and Operating Procedures," Los Alamos Scientific Laboratory report LA-7250-M (May 1978).

2. G. W. Eccleston, R. G. Schrandt, J. L. MacDonald, and F. H. Cverna "Monte Carlo Calculational Design of an NDA Instrument for the Assay of Waste Products from High Enriched Uranium Spent Fuels," Measurement Technology for Safeguards and Materials Control, Proceedings from American Nuclear Society Topical Meeting, Kiawah Island, South Carolina, November 26-30, 1979, NBS Special Publication 582.

3. G. W. Eccleston, H. O. Menlove, L. Decker, and M. W. Echo, "An Assay System to Measure High Enriched Uranium Spent Fuel and Hot Solid Waste at the Idaho FAST Facility," American Institute of Chemical Engineers Annual Meeting, Orlando, Florida, February 28, 1982.

4. "Progress Report on FAST," American Nuclear Society Nuclear News, p. 44 (July 1982).

5. LASL Group X-6, "MCNP - A General Monte Carlo Code for Neutron and Photon Transport," Los Alamos Scientific Laboratory report LA-7396-M (July 1978).

6. G. W. Eccleston, R. G. Schrandt, and J. L. MacDonald, "MCNP Modified to Provide Fission Rates by Isotope, Delayed Neutron Production, and ^{3}He Captures," Los Alamos Scientific Laboratory Group Q-1 memorandum Q-1-79-357, (1979).